TVA

Tennessee Valley Authority, Post Office Box 2000, Soddy Daisy, Tennessee 37384-2000

November 24, 2015

10 CFR 50.73

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Unit 1
Renewed Facility Operating License No. DPR-77
NRC Docket No. 50-327

Subject: **Licensee Event Report 50-327/2015-001-01, "Automatic Reactor Trip due to Negative Rate Trip as a Result of a Dropped Control Rod"**

Reference: TVA Letter submitted to NRC dated May 11, 2015, "Licensee Event Report 50-327/2015-001-00, "Automatic Reactor Trip due to Negative Rate Trip as a Result of a Dropped Control Rod."

The enclosed Licensee Event Report has been revised with supplemental information concerning the automatic reactor trip due to negative rate trip as a result of a dropped control rod. This revised report reflects the results of the root cause analysis along with corrective actions to prevent recurrence. This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv)(A), as an event that resulted in a manual or automatic actuation of the Reactor Protection System and the Auxiliary Feedwater System. Changes to the reference report are indicated by revision bars on the right side margin of the page.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact Jon Johnson, Sequoyah Acting Site Licensing Manager, at (423) 843-8129.

Respectfully,

John T. Carlin
Site Vice President
Sequoyah Nuclear Plant

Enclosure: Licensee Event Report 50-327/2015-001-01

cc: NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

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NRR

NNNRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Sequoyah Nuclear Plant Unit 1	05000327	1 OF 7

4. TITLE

Automatic Reactor Trip due to Negative Rate Trip as a Result of a Dropped Control Rod

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
03	11	2015	2015 -	001	- 01	05	11	2015	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	X **50.73(a)(2)(iv)(A)**	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
99	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Donald V. Goodin	423-843-6651

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
D	AA	CON	P430	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On March 11, 2015, at 0621 Eastern Daylight Time Sequoyah Nuclear Plant Unit 1 reactor automatically tripped due to a Negative Rate Trip as a result of Control Bank D Control Rod H-8 dropping into the core. Initial investigation revealed Control Rod H-8 dropped into the core approximately one second before the reactor trip. The dropped control rod caused a rapid decrease in power which was sensed by all four nuclear instrumentation system power range channels. Reactor trip logic is two out of four channels. No power changes or control rod motion were in progress prior to the reactor trip. All safety related equipment operated as designed, all control rods fully inserted as required, and auxiliary feedwater automatically initiated as expected. Unit 1 was stabilized in hot standby following the automatic reactor trip. The cause of the reactor trip was due to Control Rod H-8 failing to maintain its commanded position. Trouble shooting was performed on the electrical components associated with Control Rod H-8. The direct cause was a compressed four-pronged pin inside a connector for the control rod drive mechanism (CRDM) circuit. The root cause was failure of a maintenance procedure to provide inspection guidance and acceptance criteria on CRDM vertical panel connections. The corrective action to prevent recurrence includes revising the maintenance procedure and periodic preventive maintenance of CRDM connections. Unit 2 was unaffected by this event.

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Sequoyah Nuclear Plant Unit 1	05000327	2015	- 001 -	01	2 OF 7

NARRATIVE

I. Plant Operating Conditions Before the Event

At the time of the event, Sequoyah Nuclear Plant (SQN) Unit 1 reactor was operating at approximately 99 percent rated thermal power (RTP). Unit 1 Turbine load was being decreased periodically in preparation for an upcoming planned outage. The condition described in this LER did not impact SQN Unit 2.

II. Description of Events

A. Event:

On March 11, 2015 at 06:21 Eastern Daylight Time (EDT), SQN Unit 1 reactor automatically tripped due to a Negative Rate Trip as a result of Control Bank D Control Rod H-8 [EIIS Code AA] dropping into the core. Investigation revealed Control Rod H-8 dropped into the core approximately one second before the reactor trip. Control Rod H-8 is located in the center of the core and is one of nine control rods in control bank D. The dropped control rod cause a rapid decrease in power which was sensed by all four nuclear instrumentation system (NIS) power range channels. The reactor trip logic is two out of four channels.

Trouble shooting was performed on the electrical components associated with Control Rod H-8. A compressed four-pronged male pin was found inside the connector for the control rod drive mechanism (CRDM) circuit. This compressed pin created an intermittent connection which removed the stationary current from the CRDM stationary coil. This resulted in the control rod dropping into the core.

All safety related equipment operated as designed, all control rods fully inserted as required, and auxiliary feedwater automatically initiated as expected. No complications were experienced during the reactor trip.

On March 11, 2015 at 0930 EDT, NRC was notified, in accordance with 10 CFR Part 50.72(b)(2)(iv)(B), due to a reactor protection system actuation and 10 CFR Part 50.72(b)(3)(iv)(A) due to a specified system actuation.

B. Status of structures, components, or systems that were inoperable at the start of the event and contributed to the event:

There were no inoperable structures, components or systems that contributed to this event.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Sequoyah Nuclear Plant Unit 1	05000	2015	- 001 -	01	3 OF 7

NARRATIVE

C. Dates and approximate times of occurrences:

Dates and Times	Description
March 11, 2015 at 06:21 EDT	Integrated Computer System (ICS) indicates that Control Bank Control Rod H-8 starts dropping. All four NIS power range detectors indicate a drop in reactor power at the same time.
	A power range, neutron flux, negative rate trip is generated due to a rapid drop in reactor power. The remaining 52 control rods insert into the core.
	Operations perform immediate actions associated with procedure E-0, Reactor Trip or Safety Injection.
09:30	Control Rod H-8 stationary gripper fuses and blown fuse indicator are checked. No problems are identified.
1600	Continuity check performed on Control Rod H-8 coil and associated components with satisfactory results.
March 12, 2015 at 1131	Testing is performed on Control Rod H-8 to verify proper operation.
1156	Testing is performed on Control Band D Group 2 Control Rods with satisfactory results.

D. Manufacturer and model number of each component that failed during the event:

The compressed four-pronged male pin was inside the connector [EIIS Code CON] for the control rod drive circuit. The connectors are manufactured by Pyle National Company and were installed during original plant construction. The part number associated with the connector is NS2-B1720-646PN-F.

E. Other systems or secondary functions affected:

There were no other systems or functions affected by this event.

F. Method of discovery of each component or system failure or procedural error:

Reactor and turbine trip alarms annunciated alerting operators to the start of the event.

NARRATIVE

G. The failure mode, mechanism, and effect of each failed component, if known:

The compressed male pin created an intermittent connection with the female connector. This intermittent connection removed the limited stationary current from the CRDM stationary coil directly causing the control rod to drop into the core.

H. Operator actions:

The operators entered Emergency Procedure E-0, Reactor Trip or Safety Injection, and then transitioned from E-0 to Emergency Subprocedure ES-0.1, Reactor Trip Response. There were no identified complications or human performance issues associated with the trip response. Problem Evaluation Report (PER) 997605 was initiated to document that Unit 1 had tripped and start the investigation for the cause of the trip.

I. Automatically and manually initiated safety system responses:

Following the reactor trip, all plant safety systems responded as designed. All rods fully inserted as required. Auxiliary feedwater (AFW) automatically initiated from the feedwater isolation signal as expected.

III. Cause of the event

A. The cause of each component or system failure or personnel error, if known:

The direct cause was determined to be a compressed four-pronged male pin found inside a connector for the control rod drive circuit. Compression of the pin was most likely due to repetitive disconnections and reconnections of the connector over time. Degradation of the pin was not discovered prior to this event due to insufficient inspection guidance to validate the pins were making a connection.

B. The cause(s) and circumstances for each human performance related root cause:

The root cause was determined to be inadequate inspection guidance and acceptance criteria on vertical panel connections within Maintenance Instruction MI-10.29, Inspection, Cleaning and Reconnection of CRDM and RPI Connectors.

The root cause analysis is documented in PER 997605.

IV. Analysis of the event:

Prior to the event, SQN Unit 1 was operating in MODE 1 at approximately 99 percent RTP with the Reactor Coolant System (RCS) [EIIS Code AB] pressure and temperature near the nominal value of approximately 2235 pounds per square inch gauge (psig) and approximately 578 degrees Fahrenheit (F). Both the motor driven and the turbine driven auxiliary feedwater (AFW) [EIIS Code BA] pumps and steam dump valves (SDV) and the atmospheric relief valves (ARV) were available.

NARRATIVE

Following the reactor trip, RCS pressure rapidly decreased due to the decreasing RCS average temperature and the associated shrinking of coolant volume. The minimum RCS pressure was approximately 2019 psig, well above the pressure that would have initiated a safety injection signal (1870 psig). Pressurizer [EIIS Code AB] pressure recovered gradually before dropping back to normal operating pressure.

As heat removal from the steam generators (SG) [EIIS Code AB] decreased as a result of the increased steam pressure, the decrease in RCS temperature slowed and the rate of coolant shrinkage decreased. This allowed operation of the pressurizer heaters to restore RCS pressure to its nominal value. Because the maximum RCS pressure was only slightly above its nominal value following the reactor trip, pressurizer safety relief valves and power operated relief valves [EIIS AB] did not actuate.

The DNB limit for RCS average temperature of less than or equal to 583 degrees F was not exceeded. The loss of nuclear heat generation resulted in a decrease in RCS temperature to approximately 538 degrees F.

The reactor coolant pumps (RCP) [EIIS Code AB] were in service at all times during the transient and forced flow was maintained with no anomalies noted.

Prior to the trip, PZR level was being maintained in the normal program band of approximately 60 percent. Following the trip, pressurizer level followed the RCS temperature response; increasing and decreasing in magnitude, slope and duration with the RCS temperature. The minimum PZR level following the trip was approximately 22 percent. Over a 15 minute period, pressurizer level stabilized nears its program value.

The main feedwater flow rate was at nominal full power value prior to the reactor trip. When RCS average temperature dropped below 550 degrees F, main feedwater was isolated [EIIS code SJ]. The AFW system was initiated following the reactor trip on SG low-low level. AFW flow was reduced at 6 minutes after the trip to less than approximately 215 gallons per minute (gpm) to mitigate the decrease in RCS average temperature and also due to recovering SG levels.

The plant responded as expected for the conditions of the trip.

V. Assessment of Safety Consequences

There were no safety consequences as a result of the event. All safety systems functioned as designed and no complications were experienced. No Technical Specification limits were exceeded and the Updated Final Safety Analysis Report (UFSAR) analyses of the event remained bounding.

A. Availability of systems or components that could have performed the same function as the components and systems that failed during the event:

None.

NARRATIVE

B. For events that occurred when the reactor was shut down, availability of systems or components needed to shutdown the reactor and maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident:

This event did not occur when the reactor was shut down. Safety-related systems that were needed to shut down the reactor, maintain safe shutdown conditions, remove residual heat or mitigate the consequences of an accident remained available throughout the event.

C. For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from discovery of the failure until the train was returned to service:

There was no failure that rendered a train of a safety system inoperable during this event.

VI. Corrective Actions

Corrective Actions are being managed by TVA's corrective action program under PER 997605.

A. Immediate Corrective Actions:

- Reactor trip recovery completed and Unit 1 returned to service.
- Initial troubleshooting of fuses and circuit cards associated with Control Rod H-8 performed with no issues identified.
- Westinghouse troubleshooting guide steps performed with no issues identified (WCAP-15360-P, Westinghouse Rod Control Corrective Maintenance Guide).
- Control rods were successfully exercised with no issues.
- Installed testing equipment to monitor Control Rod H-8 during operation with no issues identified going into the planned refueling outage.

B. Corrective Actions to Prevent Recurrence or to reduce probability of similar events occurring in the future:

- Revise MI-10.29 to include the following:
    - Require TVA sign-off on initial CRDM and Rod Position Indicator (RPI) connector inspection step.
    - Include visual inspection guidance for male and female CRDM and RPI connections, including pin insertion depth, corrosion on pins, removed material, damaged prongs.
    - Include replacement/corrective guidance if pins do not pass visual inspection.
    - Require TVA sign-off on installation inspection of CRDM and RPI cables step.
    - Include requirement to take photographs of suspect connectors.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Sequoyah Nuclear Plant Unit 1	05000	2015	- 001 -	01	7 OF 7

NARRATIVE

- Implement periodic preventive maintenance for dimensional checks on Unit 1 and Unit 2 CRDM vertical panel connections and include the following:
  - Require quantifiable measurement on pronged pins to verify connection.
  - Include minimum acceptance criteria.
  - Include repair or contingency planning for possible replacement.

  Include guidance to verify pin retention.

VII. Additional Information

A. Previous similar events at the same plant:

A review of previous reportable events for the past three years identified six events involving inadequate procedures. LER 2-2012-001 involved an automatic reactor trip on loss of flow due to a reactor coolant pump trip. The root cause was determined to be a lack of guidance in the PM instructions for replacement of the ground fault relay that caused the trip, which had reached the end of its service life. LER 1-2013-004-01 involved a failure to comply with TSs for containment penetrations during fuel movement resulting from ineffective procedures. LER 1-2014-001-00 involved a never performed TS surveillance for the Common Spare Component Cooling System (CCS) Pump due to lack of procedural guidance. LER 1-2014-002-00 involved inadequate revision to a surveillance instruction following a Technical Specification change. LER 2-2014-002-00 involved procedures not specifying an accurate drawing for reassembling the containment vacuum relief valve and also an inadequate operating instruction for reestablishing containment integrity. LER 2-2015-001 involved an automatic reactor trip due to the failure of the main generator C-phase neutral current transformer cable. The root cause was determined to be a lack of inspections in the PM procedure to identify potential failure mechanisms.

B. Additional Information:

None.

C. Safety System Functional Failure Consideration:

This event did not result in a safety system functional failure.

D. Scrams with Complications Consideration:

This event did not result in an unplanned scram with complications.

VIII. Commitments:

None.